UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO A LICENSE AMENDMENT REQUEST

FOR A RISK-INFORMED RESOLUTION TO GSI-191

AMENDMENT NO. 206 TO FACILITY OPERATING LICENSE NO. NPF-68

AND AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NO. NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By application dated August 17, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20230A346), as supplemented by letters dated December 17, 2020, and February 15, 2021 (ADAMS Accession Nos. ML20352A228 and ML21046A094, respectively), Southern Nuclear Operating Company, Inc. (SNC, the licensee) requested changes to the technical specifications (TSs) for the Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle).

The proposed amendments would consist of changes to the License, TSs, and Final Safety Analysis Report (FSAR). The proposed amendments revise the Vogtle licensing basis as described in the Vogtle FSAR to allow the use of a risk-informed approach to the resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR [Pressurized Water Reactor] Sump Performance." The TS changes follow the model application in Technical Specification Task Force (TSTF)-567, Revision 1, "Add Containment Sump TS to Address GSI-191 Issues," (ADAMS Accession No. ML17214A813). The amendments also add a new TS 3.6.7, "Containment Sump," and an Action to address the condition of the containment sump made inoperable due to containment accident generated and transported debris exceeding the analyzed limits.

The supplements dated December 17, 2020, and February 15, 2021, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published the *Federal Register* on November 3, 2020 (85 FR 69656).

Enclosure 3

Background

By letter dated April 21, 2017 (ADAMS Accession No. ML17116A098), as supplemented by letters dated July 11 and November 9, 2017, and January 2, January 9, February 6, February 12, February 21, May 23, July 10, and December 4, 2018 (ADAMS Accession Nos. ML17192A245, ML17314A014, ML18004A070, ML18009A841, ML18037B121, ML18045A094, ML18052B342, ML18143B785, ML18193B163, and ML18338A497, respectively), SNC submitted a technical report for NRC staff review regarding the use of a risk-informed approach to resolve GSI-191 at Vogtle, Units 1 and 2, and to supplement its response to NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design-Basis Accidents at Pressurized Water Reactors," dated September 13, 2004 (ADAMS Accession No. ML042360586).

The version of the technical report enclosed with the letter dated July 10, 2018 (ADAMS Accession Nos. ML18193B163 and ML18193B165), superseded the version of the technical report enclosed with the letter dated April 21, 2017, as supplemented by letters dated July 11, 2017, and November 9, 2017, and January 2, January 9, February 6, February 12, February 21, and May 23, 2018. The version of the technical report enclosed with SNC's letter dated July 10, 2018, incorporated clarifications identified during the NRC audit and review process, and incorporated SNC responses to NRC requests for additional information.

The NRC staff found that the technical report enclosed with the letter dated July 10, 2018, is acceptable for use in plant-specific licensing applications for Vogtle, Units 1 and 2, in accordance with the limitations and conditions section and applicability provided in the NRC staff evaluation, dated September 30, 2019.

By letter dated September 30, 2019 (ADAMS Accession No. ML19120A469), the NRC issued the "Final Staff Evaluation for Vogtle Electric Generating Plant, Units 1 and 2, Systematic Risk-Informed Assessment of Debris Technical Report (EPID L-2017-TOP-0038)."

The NRC staff evaluation in the September 30, 2019, letter provided the basis for the NRC to consider use of the technical report for Vogtle in future plant-specific licensing applications in accordance with the limitations and conditions. Except for downstream effects - fuel and vessel, and licensing basis, the NRC staff concluded that the technical report contains sufficient information to address the information requested in NRC GL 2004-02. The NRC staff evaluation, dated September 30, 2019, applies only to material provided in the technical report. License amendment requests (LARs) that deviate from this technical report are subject to additional review in accordance with applicable review standards.

On November 4, 2019 (ADAMS Accession No. ML19310D797), the NRC staff held a Category 1 public meeting (currently classified as an Observation meeting) to clarify the NRC staff evaluation dated September 30, 2019.

On May 14, 2020 (ADAMS Accession No. ML20136A248), the NRC staff held a Category 1 public pre-submittal meeting at the request of SNC. Once again, the NRC staff reviewed limitations and conditions in the NRC staff evaluation dated September 30, 2019.

System Descriptions and Historical Information

The function of the emergency core cooling system (ECCS) is to cool the reactor core and provide shutdown capability following a loss-of-coolant accident (LOCA). The primary functions of the containment spray system (CSS) are to reduce containment pressure and reduce the concentration and quantity of fission products in the containment building after a LOCA. Nuclear plants are designed and licensed with the expectation that they are able to remove reactor decay heat following a LOCA to prevent core damage. Long-term core cooling (LTCC) following a LOCA is also a basic safety function for nuclear reactors.

As part of the actions to resolve GSI-191, in September 2004, the NRC issued GL 2004-02 to holders of operating licenses for PWRs. The GL requested that licensees provide information that demonstrated that debris would not interfere with the operation of the ECCS and CSS systems during the recirculation phase of a LOCA response.

The Commission issued SRM-SECY-12-0093 on December 14, 2012 (ADAMS Accession No. ML12349A378), approving three options for closure of GSI-191. These options are described in the Attachment to this safety evaluation (SE).

By letter dated May 16, 2013 (ADAMS Accession No. ML13137A130), SNC stated that it would pursue Option 2 for the closure of GSI-191 and GL 2004-02, and intended to use a risk-informed methodology.

For further details on the system descriptions and history, please see the Attachment to this SE.

Description of Planned Sump Strainer Modification

The licensee's risk-informed analysis, completed after the installation of the residual heat removal (RHR) screens, led to a proposal to modify the screens to reduce their height by removing two disks per stack. This allows the RHR screens to be fully submerged for an increased number of postulated scenarios. The RHR strainer assemblies are described in the Attachment to this SE. The strainers will be the same as described except that each stack will consist of 16 disks (instead of 18 disks) resulting in a perforated plate surface area of about 678 ft^2 and circumscribed area of about 159 ft^2 per sump. The 16-disk strainer is about 48 inches tall and the elevation of the top of the upper strainer disk is about 53.25 inches. The licensee's analyses in its technical report and LAR, and the associated staff evaluations, are based on the 16-disk strainer configuration. The CS strainers are not being modified because they are shorter than the RHR strainers and are submerged before the RHR strainers in all scenarios.

Current TSs

The current Vogtle TS pages: Table of Contents; TS 3.5.2, "ECCS [Emergency Core Cooling System] – Operating;" and TS 3.5.3, "ECCS – Shutdown" are contained in ADAMS Accession No. ML052840233.

Proposed TS Changes

By letter dated August 17, 2020, SNC proposed changes to Vogtle TSs. The changes incorporate TSTF-567, Revision 1. The NRC issued a final SE approving TSTF-567, Revision 1, on July 3, 2018 (ADAMS Accession No. ML18116A606). SNC proposed changes to pages iii, 3.5.2-3, 3.5.3-2, and 3.6.7-1, as well as the addition of a new TS page 3.6.7-2. The description of these changes is contained in Enclosure 3, Attachments 1 and 2 to the proposed LAR. The licensee provided supplements by letters dated December 17, 2020, and February 15, 2021. These letters did not affect the requested changes to the TS related to its proposed TSTF-567 adoption.

Accordingly, the amendment would revise TS 3.5.2, and TS 3.5.3. The proposed changes would also add a new TS, “Containment Sump,” to Section 3.6, “Containment Systems.” Although, the proposed changes are based on TSTF-567, the licensee proposed several variations from the TS changes described in TSTF-567. These variations are described in Section 2.4.4 of this SE and evaluated in Section 3.2.4.

Proposed FSAR Changes

By letter dated August 17, 2020, SNC proposed changes to Vogtle FSAR contained in Enclosure 2, Attachment 1 of its LAR. These changes describe the treatment of debris with respect to operation of the ECCS and CSS during sump recirculation.

2.0 REGULATORY EVALUATION

2.1 Risk-Informed Resolution of GL 2004-02

The NRC staff's acceptance criteria for ECCS performance following a LOCA are based on Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.” LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the reactor coolant pressure boundary at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate these accidents. The NRC staff’s review covered the acceptance criteria based on 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria, considering the effects of debris as specified in GL 2004-02.

The following regulatory requirements are applicable to the review.

Section 50.46(a)(1)(i) of 10 CFR states, in part, that each PWR to be provided with an ECCS, and the ECCS performance must be calculated with an acceptable evaluation model and must be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated.

Section 50.46(b)(5) of 10 CFR states, in part, that the calculated core temperature maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Section 50.46(c)(2) of 10 CFR, states in part, that an evaluation model -is the calculational framework for evaluating the behavior of the reactor system during a postulated LOCA. It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as the mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in the computer programs, values of parameters, and all other information necessary to specify the calculational procedure. Although not traditionally considered as a component of the 10 CFR 50.46 ECCS evaluation model, the calculation of sump performance is necessary to determine if the sump and the ECCS are predicted to provide enough flow to ensure long-term cooling.

Applicable Regulatory Guides, Review Plans, and Guidance Documents

The Nuclear Energy Institute (NEI) developed an evaluation guidance document entitled "PWR Containment Sump Evaluation Methodology," dated May 28, 2004 (ADAMS Accession No. ML041550661), for use by the industry. On December 6, 2004, the NRC issued an SE that found the NEI document provided an acceptable overall guidance methodology, but noted that portions needed additional justification and modification (ADAMS Accession No. ML043280641). Modifications were made, and the final guidance was provided as NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," in December 2004 (ADAMS Accession Nos. ML050550138 and ML050550156). Together, Volume 1 of NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," dated December 2004, and Volume 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02," dated December 6, 2004 (ADAMS Accession No. ML042360586), describe a method acceptable to the NRC staff, with limitations and conditions for performing the evaluations requested by GL 2004-02.

In addition to the evaluation guidance of NEI 04-07, the industry developed the following topical reports (TRs) to aid licensees in responding to GL 2004-02.

- TR-WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," March 2008 (ADAMS Accession No. ML081150379).

- TR-WCAP-16406-P-A, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1, March 2008 (ADAMS Accession No. ML081000025).

- TR-WCAP-16793-NP-A, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," Revision 2, July 2013 (ADAMS Accession No. ML13239A114).

The reports listed above, subject to the limitations and conditions contained in the NRC SEs for those TRs, describe methods acceptable to the NRC staff for performing the evaluations and analyses within the scope stated in those documents.

To more clearly communicate the NRC staff's expectations for the level of technical detail in the licensees' submittals, the NRC staff issued documents entitled "Revised Content Guide for Generic Letter 2004-02 Supplemental Responses," dated November 21, 2007 (ADAMS Accession No. ML073110389), and "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized Water Reactors,'" dated March 28, 2008

(ADAMS Accession No. ML080230234). The content guide describes the information expected to be addressed in the NRC review in each of the following areas:

- corrective actions taken to address GL 2004-02
- break selection
- debris generation and zone of influence
- debris characteristics
- latent debris
- debris transport
- head loss and vortexing
- net positive suction head
- coatings evaluation
- debris source term
- screen modification package
- sump structural analysis
- upstream effects
- downstream effects - components and systems
- downstream effects - fuel and vessel
- chemical effects
- licensing basis

The NRC staff later issued staff guidance for the review of in-vessel effects on September 4, 2019, “NRC Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses” (ADAMS Accession No. ML19228A011). The staff review guidance is based on a staff “Technical Evaluation Report (TER) Of In-Vessel Debris Effects” issued June 13, 2019, (ADAMS Accession No. ML19073A044, non-publicly available). The TER relies on a significant amount of information from a PWROG [Pressurized Water Reactor Owners Group] TR WCAP-17788 issued on July 17, 2015 (ADAMS Package Accession No. ML15210A667).

NUREG-0800, Section 3.8.3, Revision 4, “Concrete and Steel Internal Structures of Steel or Concrete Containments,” of the Standard Review Plan (SRP) (ADAMS Accession No. ML13198A250), September 2013, lists acceptable codes and standards for design of containment internal structures.

Regulatory Guide (RG) 1.82, Revision 4, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” March 2012 (ADAMS Accession No. ML111330278), provides guidance for an evaluation of the effects of debris on ECCS strainers and, more generally, guidance for the evaluation of water sources for long-term recirculation following a LOCA.

RG 1.174, Revision 3, “An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” January 2018, (ADAMS Accession No. ML17317A256), provides guidance on the use of probabilistic risk assessment (PRA) findings and risk insights in support of licensee requests for plant-specific changes to a licensing basis. This RG provides risk acceptance guidelines for evaluating the results of such evaluations. RG 1.174 also provides the five key principles of risk-informed integrated decision-making.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ADAMS Accession No. ML090410014), endorses, with clarifications, the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) PRA Standard ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S 2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (ASME/ANS 2009 Standard). The ASME/ANS 2009 Standard addresses PRAs for internal events and other hazards. RG 1.200 describes one acceptable approach for determining whether the technical adequacy of the PRA, in total, or the parts that are used to support an application, is acceptable for use in regulatory decision-making for light-water reactors.

SRP Chapter 15, Section 15.0.2, "Review of Transient and Accident Analysis Methods," (ADAMS Accession No. ML053550265) and SRP Chapter 19, Section 19.1, Revision 3, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load," (ADAMS Accession No. ML062510220) also provide for evaluating engineering issues and PRA acceptability.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in NUREG-0800, SRP Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (ADAMS Accession No. ML071700658). Section 19.2 of this SRP references the same criteria as RG 1.174, and states that a risk-informed application should be evaluated to ensure that the proposed changes meet the five key principles of risk-informed decision-making.

2. 2 Applicable Regulatory Requirements and Guidance for Implementation of TSTF-567

2.2.1 Technical Specification Requirements

Section 50.36(a)(1) of 10 CFR states, in part, that each applicant for a license authorizing operation of a production or utilization facility shall include in its application proposed TSs. That regulation also states, in part, that "[a] summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications."

The regulations in 10 CFR 50.36(b), state that each license authorizing operation of a production or utilization facility will include TS and that the TS will be derived from the analyses and evaluations included in the safety analysis report, and amendments thereto, submitted in accordance with 10 CFR 50.34.

The regulations in 10 CFR 50.36(c), states that TS will include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

The regulations in 10 CFR 50.36(c)(2)(i), state that LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility and that, when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. The regulations in 10 CFR 50.36(c)(3) states that surveillance requirements (SRs) are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

The regulations in 10 CFR 50.36(c)(5) state that administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner."

2.2.2 Guidance

The guidance that the NRC staff considered in its review of this LAR included the following:

- NUREG-0800, Revision 3, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Chapter 16.0, "Technical Specifications," March 2010 (ADAMS Accession No. ML100351425), provides guidance on review of TSs.

- NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 4, Volume 1, "Specifications," and Volume 2, "Bases," April 2012 (ADAMS Accession Nos. ML12100A222 and ML12100A228, respectively).

2.3 TS Changes for Implementation Of TSTF-567

The TSs include limiting conditions for operation (LCOs), which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Specified with each stated condition of the LCO are required action(s) and completion time(s) (CTs) to meet TS requirements.

2.3.1 TS 3.5.2, "ECCS - Operating"

The function of the ECCS is to provide core cooling and negative reactivity to ensure the reactor core is protected after any of the following accidents:

a. Loss-of-coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system,

b. Rod ejection accident,

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater, and

d. Steam generator tube rupture.

TS 3.5.2 is applicable in Modes 1, 2, and 3 and requires that two independent ECCS trains be operable to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train.

TS 3.5.2 helps ensure the following acceptance criteria for ECCS, established by 10 CFR 50.46, will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200 degrees Fahrenheit (°F),

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation,

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,

d. Core is maintained in a coolable geometry, and

e. Adequate long-term core cooling capability is maintained.

TS 3.5.2 also limits the potential for a post-trip return to power following a main steam line break event and ensures that containment temperature limits are met.

2.3.2 <u>TS 3.5.3, "ECCS - Shutdown"</u>

TS 3.5.3 is applicable in Mode 4 and requires one of the two independent (and redundant) ECCS trains to be operable to ensure that sufficient ECCS flow is available to the core following a design-basis accident.

2.4 <u>Proposed Changes to the TSs for Implementaion of TSTF-567</u>

The proposed changes would revise TS 3.5.2, "ECCS - Operating", and TS 3.5.3, "ECCS – Shutdown." The proposed changes would also add a new TS, "Containment Sump" to Section 3.6, "Containment Systems." The proposed changes are described below.

2.4.1 <u>Proposed Changes to TS 3.5.2, "ECCS - Operating"</u>

TS 3.5.2 currently contains Surveillance Requirement (SR) 3.5.2.7, which requires the following at a frequency in accordance with the Surveillance Frequency Control Program (SFCP):

> Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.

The licensee proposed to modify and move SR 3.5.2.7 from TS 3.5.2 and include it in the new TS 3.6.7, "Containment Sump."

This change is evaluated in Section 3.2.1 of this SE.

2.4.2 <u>Proposed Changes to TS 3.5.3, "ECCS - Shutdown"</u>

TS 3.5.3 currently contains SR 3.5.3.1 which cites applicable SRs required under TS 3.5.2 for all equipment to be OPERABLE. One of those referenced SRs is SR 3.5.2.7, as described in Section 2.4.1 of this SE.

Because the licensee proposed to modify and move SR 3.5.2.7 from TS 3.5.2 and include applicable provisions in the new containment sump TS 3.6.7, the licensee proposed to delete the applicability requirement of SR 3.5.2.7 in SR 3.5.3.1.

This change is evaluated in Section 3.2.2 of this SE.

2.4.3 Proposed New 3.6.7, "Containment Sump"

Proposed new TS 3.6.7 would require four containment sumps to be operable in Modes 1, 2, 3, and 4. Condition A would state, "One or more containment sumps inoperable due to containment accident generated and transported debris exceeding the analyzed limits," then the licensee is required to: (1) initiate action to mitigate the containment accident generated and transported debris immediately, (2) perform SR 3.4.13.1 once per 24 hours, and (3) restore the containment sump to OPERABLE status within 90 days (Required Actions A.1, A.2, and A.3, respectively). Surveillance Requirement 3.4.13.1 requires verification that the reactor coolant system (RCS) operational leakage is within limits by performance of an RCS water inventory balance.

Condition B, Required Actions B.1 and B.2 specify that if the containment sumps are inoperable for reasons other than Condition A, then the licensee is required to immediately declare the affected ECCS and CSS trains inoperable. This is a variation from TSTF-567. TSTF-567 requires notes for Required Action B.1. Vogtle's variation is a simplification of the TSTF-567 conditions and is implemented because each Vogtle unit has a separate strainer for each pump that receives flow from the emergency sump.

Condition C specifies that if required actions and associated CTs under Condition A are not met, then the licensee is required to be in Mode 3 in 6 hours and Mode 5 in 36 hours (Required Actions C.1 and C.2, respectively). This is a variation from TSTF-567. TSTF-567 proposes the same actions and completion times if Condition B is not met. For Vogtle, Condition C applies only to Condition A.

The licensee proposed to modify and move SR 3.5.2.7 currently located in TS 3.5.2. New SR 3.6.7.1 would require the licensee to verify, by visual inspection, the containment sump does not show structural damage, abnormal corrosion, or debris blockage in accordance with the SFCP.

The containment sump design for Vogtle includes more than one containment sump. Each unit has four strainers, two for the CSS pumps and two for the ECCS pumps. Each strainer provides flow to its individual pump. For Condition A, the sumps are considered part of a single support system because containment accident generated and transported debris issues that would render one sump inoperable would render all of the sumps inoperable. The new containment sump TS proposed is applicable to plants that have more than one containment sump.

The licensee also proposed a conforming administrative change to the TS Table of Contents to reflect the addition of the new containment sump TS.

This change is evaluated in Section 3.2.3 of this SE.

2.4.4 Proposed Variations from TSTF-567, Revision 1

The licensee is proposing the following variations from the TS changes described in TSTF-567 or the applicable parts of the NRC staff's SE of TSTF-567. These variations do not affect the applicability of TSTF-567 or the NRC staff's SE to the proposed LAR.

The Vogtle TSs utilize different numbering than the Standard Technical Specifications (STS) on which TSTF-567 was based. Specifically, SR 3.5.2.8 in NUREG-1431 is SR 3.5.2.7 in the Vogtle TSs and the new containment sump specification, TS 3.6.19, in TSTF-567 is TS 3.6.7 in the Vogtle TSs. These differences are administrative in nature and do not affect the applicability of TSTF-567 to the proposed LAR. This SE uses the Vogtle specific TS numbers throughout.

The Vogtle TSs contain a SFCP. Consequently, the SR Frequency for SR 3.6.7.1 would continue to be "In accordance with the Surveillance Frequency Control Program."

As discussed above, the licensee proposed Required Actions for Condition B different than those specified in TSTF-567. The proposal eliminates the traveler Notes that require the affected unit to enter the applicable Conditions and Required Actions of LCOs 3.5.2 and 3.5.3, and 3.6.6 for trains made inoperable by the containment sumps. The proposal also changes the CT from the TSTF, which allows 72 hours (or in accordance with the risk-informed completion time) to restore the sumps to operable status. Instead, the licensee proposed Required Actions (B.1 and B.2) to declare the affected ECCS and CSS trains inoperable in accordance with LCOs 3.5.2 and 3.5.3, and 3.6.6. The licensee proposed that these actions have a CT of immediately. This simplification is possible because each CSS and ECCS pump has its own strainer and sump.

The licensee also proposed to make Condition C applicable only if the Required Actions and CTs of Condition A are not met. TSTF-567 applies Condition C if the Condition B Required Actions and CTs are not met. Condition C requires that the unit be placed in Mode 3 in 6 hours and Mode 5 in 36 hours if the required actions and completion times are not met. For Vogtle, Condition C is proposed to apply only if the Condition A Required Actions are not met within the CT. The immediate implementation of Required Actions B.1 and B.2 make it unnecessary to apply Condition C to Condition B.

3.0 TECHNICAL EVALUATION

3.1 Risk-Informed Resolution of Generic Letter 2004-02

As discussed in the Background portion of Section 1.0 above, the NRC staff previously reviewed a technical report submitted by the licensee. The NRC found SNC's the technical report (ADAMS Accession Nos. ML18193B163 and ML18193B165) acceptable for use in plant-specific licensing applications for Vogtle, Units 1 and 2. The NRC staff's conclusions are documented in a final staff evaluation of the Vogtle technical report dated September 30, 2019 (ADAMS Accession No. ML19120A469). The NRC staff evaluation stated that future licensing actions are contingent on addressing the limitations and conditions. Any deviation from the Vogtle technical report will be subject to additional review in this SE.

The limitations and conditions identified by the NRC staff in its staff evaluation are as follows:

1. The applicability of the NRC's acceptance is limited to the structures, systems, and components; plant configurations; and operations described in Enclosures 2, 3, and 4 of SNC's letter dated July 10, 2018 and the strainer design described in the Section entitled, "16-Disk ECCS Suction Strainer Summary," of Enclosure 2.
2. The applicability of the NRC's acceptance is limited to the Vogtle assessment of risk attributable to debris described in Enclosures 1 and 3 of SNC's letter dated July 10, 2018.

3. Describe in-vessel analysis, establish in-vessel acceptance criteria, and demonstrate the criteria are met.
4. Address Key Principle 1 (i.e., the proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption) and Key Principle 5 (i.e., the impact of the proposed licensing basis change should be monitored using performance measurement strategies) in RG 1.174, Revision 3.
5. Identify key elements of the risk-informed analysis (e.g., methods, approaches, and data) that will be described in the Vogtle UFSAR.
6. Identify key elements of the risk-informed analysis and corresponding methods, approaches, and data that, if changed, would constitute a departure from the method used in the safety analysis as defined by 10 CFR 50.59.
7. Identify the relevant elements of the risk-informed assessment that may need to be periodically updated. The licensee must describe the program or controls that will be used to ensure relevant elements of the risk-informed assessment are periodically updated.
8. Describe a reporting and corrective action strategy for addressing situations in which an update to the risk-informed assessment reveals that the acceptance guidelines described in Section 2.4 of RG 1.174, Revision 3, have been exceeded.
9. Correct the error concerning the evaluation of transported coatings debris loads described in SNC's letter dated December 4, 2018. Specifically, provide corrected coating debris volumes and describe how coating debris loads on the strainers are determined. In addition:
    a. Verify that the use of the corrected coating debris volumes has a limited impact on strainer head loss and the head loss is acceptable. Also, the licensee must describe the method of verification.
    b. Verify that the use of the corrected coating debris volumes has a limited impact on CDF [core damage frequency] and does not result in exceeding the acceptance guidelines for very small change in risk, as described in Section 2.4 of RG 1.174, Revision 3. Also, the licensee must describe the method of verification.

The limitations and conditions are evaluated in this section.

3.1.1 Limitation and Condition 1

The applicability of the NRC's acceptance is limited to the structures, systems, and components; plant configurations; and operations described in Enclosures 2, 3, and 4 of SNC's letter dated July 10, 2018, and the strainer design described in the Section entitled, “16-Disk ECCS Suction Strainer Summary,” of Enclosure 2 of the 2018 letter.

3.1.1.1 NRC Staff Evaluation

The NRC staff has reviewed the LAR and finds that that the information provided is consistent with the licensee’s description included in its letter dated July 10, 2018. The NRC staff understands that the RHR strainers have not yet been reduced to 16 disks in height (by removing the top two disks of each strainer) and that this work will be performed following approval, but prior to implementation of the license amendment as stated in letter dated August 17, 2020. The NRC staff concludes that the proposed LAR is sufficient to address Limitation and Condition 1 and that conforming changes are planned to modify the RHR strainer height.

Based on the above, the NRC staff finds the licensee's proposed actions to address Limitation and Condition 1 are acceptable. In its letter dated August 17, 2020, the licensee states:

> This license amendment is effective as of its date of issuance and shall be implemented after the strainers have been modified consistent with the design described in Enclosure 2 of SNC's letter dated July 10, 2018, for Vogtle, Units 1 and 2, and prior to the conclusion of the Vogtle, Unit 2, Spring 2022 refueling outage.

3.1.2 Limitation and Condition 2

The applicability of the NRC's acceptance is limited to the Vogtle assessment of risk attributable to debris described in Enclosures 1 and 3 of SNC's letter dated July 10, 2018.

3.1.2.1 NRC Staff Evaluation

The NRC staff has reviewed the LAR and finds that the information provided is consistent with the licensee's description included in its letter dated July 10, 2018. The licensee's resolution of Limitation and Condition 9 (see section 3.1.9 below) resulted in a very small increase in the risk. Based on its review of the risk assessment results in the LAR, the NRC staff concludes that (1) the licensee used the NRC-accepted methodology described in Enclosures 1 and 3 of SNC's letter dated July 10, 2018 for the risk-informed assessment, and (2) the updated results do not exceed the RG 1.174 Region III risk acceptance guidelines. Based on the above, the NRC staff finds that the licensee's justification for Limitation and Condition 2 is acceptable, because the methodology remained unchanged, the as-built and as-operated plant is modeled, and the results continue to meet the RG 1.174 Region III risk acceptance guidelines.

3.1.3 Limitation and Condition 3

Describe in-vessel analysis, establish in-vessel acceptance criteria, and demonstrate the criteria are met.

3.1.3.1 In-vessel Evaluation Technical Description

Because the in-vessel analysis was not evaluated previously by the NRC, a more detailed evaluation is provided in this section of the SE. The NRC's evaluation focuses on the hot-leg break because the NRC guidance, "U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses" (ADAMS Accession No. ML19228A011) (GL 2004-02 NRC Staff Guidance) found that the cold-leg break did not need to be evaluated on a plant-specific basis.

The licensee stated that the amount of fibrous debris that could bypass the strainers and transport to the reactor vessel was evaluated in its July 2018 submittal. The licensee also stated that the LAR evaluation of in-vessel debris effects is based on the NRC staff review guidance and PWROG guidance on the topic. The licensee used Option 4 of the GL 2004-02 NRC Staff Guidance.

For Option 4, the NRC guidance states that the licensee should confirm that the plant-specific parameters are within the bounds for key parameters defined in the guidance. In Attachment 3 to its letter dated August 17, 2020, the licensee provided Table A3-1, “Summary of In-Vessel Effects Parameters,” to compare the bounding values with those for Vogtle.

Table A3-1 from the licensee’s submittal is reproduced below.

Parameters	**Values from WCAP-17788-P, Revision 1**	**Vogtle Values**
Nuclear Steam Supply System (NSSS) Design	Various	Westinghouse 4-loop
Fuel Type	Various	Westinghouse 17 x 17 VANTAGE 5, VANTAGE+ fuel
Barrel/Baffle Configuration	Various	Upflow
Minimum Chemical Precipitation Time	t_{block} from WCAP-17788 143 minutes	24 hours
Maximum HLSO Time	N/A	8 hours
Maximum Core Inlet Fiber Load for Hot Leg Break (HLB)	WCAP-17788, Volume 1, Table 6-3	90.61 g/FA
Total In-Vessel Fiber Limit for HLB	WCAP-17788, Volume 1, Section 6.4	N/A
Minimum Sump Switchover (SSO) Time	20 minutes	31.9 minutes
Maximum Rated Thermal Power	3658 MWt	3625.6 MWt
Maximum Alternate Flow Path (AFP) Resistance	WCAP-17788, Volume 4, Table 6-1	WCAP-17788, Volume 4, Table RAI-4.2-24
ECCS Flow per FA	8 – 40 gpm/FA	15.5 gpm/FA

The table contains references to WCAP-17788 in place of proprietary information. The table does not contain proprietary information. The information is available in the referenced sections of WCAP-17788.

The first three rows of the table list the important parameters for the fuel and reactor vessel design configurations. The design of the fuel and vessel are inputs in the models that determine the bounding values for the remainder of the items in the table. The licensee evaluated the remaining key parameters against the bounding values from WCAP-17788 and NRC staff guidance.

The licensee stated that testing demonstrated that the earliest precipitation time for Vogtle plant conditions is 24 hours. The licensee compared this with t_{block} from WCAP-17788. "t_{block}" is the earliest time that the core inlet could become completely blocked and adequate core cooling would still occur due to flow through alternate flow paths (AFPs). If the core inlet becomes blocked prior to t_{block}, the fuel will begin to heat up. If the core is not blocked until after t_{block} adequate cooling is delivered to the core.

The licensee stated that it is unlikely for the core to block without chemicals present. The values in the chemical effects timing row show that chemicals will not contribute to core blockage before 24 hours and that after 143 minutes the design can accommodate full core inlet blockage. The comparison shows over 21 hours of margin between the time that AFPs can provide adequate flow and the earliest time that the core is likely to become fully blocked.

The licensee stated that the maximum hot-leg switchover (HLSO) time for the plant is 8 hours. That means that HLSO would be initiated within 8 hours following an event. Aligning to hot-leg injection bypasses any blockage at the core inlet by injecting coolant into the top of the core via the hot leg. This is also compared against the minimum chemical effects time of 24 hours. There is a significant margin between HLSO time and the earliest time that chemicals may form. Therefore, any core blockage is bypassed before full core blockage is likely to occur.

The LAR compared the amount of fiber calculated to reach the core inlet with the limit established for the Vogtle configuration by WCAP-17788. The licensee stated that the fiber amount that may reach the core inlet is greater than the WCAP limit for the Vogtle design. The licensee stated that the amount of fiber calculated to reach the core is less than the total in-vessel fiber limit. (The total in-vessel fiber limit is greater than the core inlet fiber limit.) The licensee stated that because any fiber buildup at the core inlet is expected to be non-uniform the head loss associated with the fiber would be lower than predicted. The lower head loss would allow additional fiber accumulation before full core blockage would occur. The licensee stated that, if the maximum amount of fiber arriving at the core is less than the WCAP limit for total in-vessel fiber for the hot-leg break, LTCC is assured.

The licensee stated that the earliest time to switch to sump recirculation (sump switchover (SSO)) at Vogtle is greater than that assumed in the WCAP analyses. Debris accumulation at the core inlet does not begin until after SSO. The WCAP analysis assumes SSO time of 20 minutes, while the minimum Vogtle SSO time is 31.9 minutes. This provides significant time for decay heat to decrease thus reducing the coolant flow required for core cooling.

The licensee stated that the plant licensed thermal power (3625.6 megawatts thermal (MWt)) is less than that assumed in the WCAP analyses (3658 MWt).

The licensee stated that the Vogtle AFP resistance is less than that assumed in the WCAP analyses. These values are proprietary and are withheld from public disclosure. However, the NRC staff obtained the values from the proprietary references and verified the licensee's assertion.

The licensee provided a comparison of the plant minimum ECCS flow rate per fuel assembly to the analysis flowrates. The Vogtle flow rate is 15.5 gallons per minute (gpm) while the analysis flowrates considered 8 to 40 gpm.

The licensee concluded, based on the comparisons, that in-vessel downstream effects would not challenge LTCC at Vogtle.

3.1.3.2 NRC Staff Evaluation

In the Debris Transport Submodel Section of the NRC staff evaluation of the licensee's July 2018 submittal, the NRC concluded that the fiber penetration and transport analysis was conducted such that a conservative amount of fiber was predicted to reach the core. The calculation of the amount of fiber that could arrive at the core was performed acceptably and is, therefore, not discussed further in this SE.

The NRC staff reviewed the licensee's response regarding the effects of debris in the reactor vessel on LTCC against the NRC staff review guidance, "U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses, issued on September 4, 2019 (ADAMS Accession No. ML19228A011). All of the key parameters for in-vessel evaluations at Vogtle are within the limits established by the staff guidance except for the core inlet fiber amount. In its TER on in-vessel debris effects, the NRC staff recognized that the core inlet fiber limit was based on uniform bed formation at the core inlet. The TER concluded that the debris deposition would be non-uniform because of non-uniform flow approaching the core inlet. This behavior was credited by the licensee in its evaluation.

As an additional check on the potential for complete core blockage, the NRC staff reviewed information provided by the licensee in its previous submittal dated July 10, 2018, and the NRC staff evaluation dated September 30, 2019. The licensee calculated the amount of fiber that could reach the core inlet by t_{block}. This amount was less than the WCAP limit for the Vogtle design. Therefore, AFPs would allow adequate coolant flow into the core before the amount of fiber that was predicted conservatively to result in complete core blockage arrives at the core inlet.

The NRC staff compared the amount of debris calculated to reach the core to the WCAP limit. The NRC staff agrees with the licensee's conclusion that the amount of fibrous debris calculated to reach the core will not block the core inlet prior to the availability of adequate cooling flow through alternate flow paths. The NRC staff conclusion considered that chemical effects will not be present until well after the AFPs can provide adequate cooling and well after the maximum time for HLSO. Fuel assembly testing has shown that in the absence of chemical effects, larger amounts of fiber can collect at the core inlet without causing complete flow blockage. Considering the licensee's estimation of the timing associated with a hypothetical, complete core inlet blockage, such a blockage is unlikely, and, if formed, would be delayed relative to the associated timing assumed in the WCAP-17788 analyses. Since the remaining key parameters are well represented in the WCAP-17788 analyses, the NRC staff determined that the licensee has demonstrated sufficiently that the WCAP-17788 analyses remain representative of Vogtle, such that the analyses can be used to demonstrate that Vogtle would maintain adequate LTCC following a complete core inlet blockage, notwithstanding the higher debris amounts that may be present at Vogtle following a postulated LOCA.

3.1.3.3 NRC Staff Conclusion

For in-vessel downstream effects, the NRC staff concludes that the licensee provided information that demonstrates that the in-vessel area has been addressed conservatively or prototypically. The NRC staff concludes that there is reasonable assurance that debris in the vessel will not inhibit LTCC. Therefore, the NRC staff finds that the licensee's evaluation is acceptable. Based on the above, the NRC staff finds sufficient basis to close this area for GL 2004-02, and that the licensee's justification for Limitation and Condition 3 is acceptable.

3.1.4 Limitation and Condition 4

Address Key Principle 1 (i.e., the proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption) and Key Principle 5 (i.e., the impact of the proposed licensing basis change should be monitored using performance measurement strategies) in RG 1.174, Revision 3.

3.1.4.1 NRC Staff Evaluation and Conclusion

In its letter dated August 17, 2020, the licensee requested an exemption under 10 CFR 50.12 from certain requirements of 10 CFR 50.46(a)(1). Section 2.0 of Enclosure 1 to the licensee's letter describes the proposed exemption request and Section 4.0 of that enclosure provides the basis for the proposed exemption. The exemption request complements the LAR provided in Enclosures 2 through 4 of the letter, dated August 17, 2020. The NRC staff concludes that the licensee has sufficiently addressed Key Principle 1 in RG 1.174, Revision 3, because the proposed licensing basis change is addressed explicitly in the requested exemption ADAMS Package Accession No. ML21071A050).

Regarding Key Principle 5, the licensee described how it will monitor sump strainer performance and assess the impact on risk. The licensee plans to utilize programs and procedures to inspect and limit the potential for debris accumulation in containment. The licensee stated that it has implemented procedures and programs for monitoring, controlling, and assessing changes to the plant that have a potential impact on plant performance related to GSI-191 concerns. The licensee further stated that these procedures and programs provide the capability to monitor the performance of the sump strainers and assess impacts to the inputs and assumptions used in the PRA and the associated engineering analysis that support the proposed change. The licensee provided a list of relevant programmatic requirements including its Maintenance Rule program, its online configuration risk management procedure, and its quality assurance program. In addition, the licensee stated that no changes are made to ASME Section XI inspection programs or mitigation strategies. The NRC staff concludes that the licensee sufficiently addressed Key Principle 5 of RG 1.174, Revision 3.

Based on the above, the NRC staff concludes that the licensee has identified a set of existing programs and procedures to monitor the performance of the sump strainers in containment and assess the impact on the risk-informed assessment, including the PRA and engineering analysis that supported the LAR in letter dated August 17, 2020. Based on the above, the NRC staff concludes that the licensee has sufficiently addressed Key Principle 5 from RG 1.174. The NRC staff concludes that the licensee's justification for Limitation and Condition 4 is acceptable.

3.1.5 Limitation and Condition 5

Identify key elements of the risk-informed analysis (e.g., methods, approaches, and data) that will be described in the Vogtle UFSAR.

3.1.5.1 Description of the Key Elements of the Risk-Informed Analysis

In its LAR, the licensee stated that the key elements of the risk-informed analysis include the following:

1. The methodology used to quantify the amount of debris generated at each break location, including the assumed zone of influence (ZOI) size based on the target destruction pressure and break size, and the assumed ZOI shape (spherical or hemispherical) based on whether the break is a DEGB [double ended guillotine break] or partial break. This requirement applies to the methodology, but not the tool or program used.
2. The methodology used to evaluate debris transport to the RHR and CSS strainers. This requirement applies to the methodology, but not the tool or program used.
3. The methodology used to quantify chemical precipitates, including the refinements to WCAP-16530-P-A, application of the solubility correlation, and application of the WCAP-17788-P autoclave testing. This requirement applies to the methodology, but not the tool or program used.
4. The strainer debris limits shown in TS Bases Table B 3.6.7-1 (see Attachment 3 of Enclosure 3 in this submittal), which are based on tested and analyzed debris quantities.
5. The methodology and acceptance criteria used to assess ex-vessel component blockage and wear.
6. The methodology used to assess in-vessel fiber accumulation and the associated limits (see Attachment 3 of this enclosure). This requirement applies to the methodology, but not the tool or program used.
7. The methodology used to quantify CFPs [conditional failure probabilities], ΔCDF [change in core damage frequency] and ΔLERF [change in large early release frequency]. This requirement applies to the methodology, but not the tool or program used.

3.1.5.2 NRC Staff Evaluation

The NRC staff reviewed the aspects of the licensee's analysis and confirmed that they were inclusive of those elements necessary to define the methodology used to evaluate the effects of debris on LTCC.

The NRC staff also reviewed the UFSAR markup included in the LAR to ensure that the elements described in the LAR were incorporated appropriately into the UFSAR. The staff noted that the UFSAR markup includes additional details describing each of the seven key elements listed above. The UFSAR markup contains adequate detail to define the methods used in the analysis.

The NRC staff noted that, as originally submitted by the licensee, both the LAR and UFSAR descriptions of the key elements excluded the tool or program used to implement the methodology. During its evaluation of the Vogtle risk-informed method, industry and the NRC staff spent considerable resources to verify that the program used (NARWHAL) provided adequate results. The NRC staff requested the licensee to provide additional information regarding the potential use of a different program to calculate the result of the risk-informed analysis. The staff expressed concern that the verification, validation, and other controls that had been implemented for the current program would be difficult to recreate in a way that assures reliable calculations. By email dated December 3, 2020 (ADAMS Accession No. ML20338A151), the NRC issued a request for additional information (RAI) on this issue. The licensee responded in letter dated December 17, 2020 (ADAMS Accession No. ML20352A228). In its response, the licensee provided revised language that includes NARWHAL as a key element of the risk-informed analysis. The response also provided new UFSAR page markups that state that the program is a key element of the analysis. The licensee acknowledged that some conservative calculations may be performed outside the NARWHAL software, but stated that integrated analyses require the use of NARWHAL. The NRC staff found that the licensee's response to the RAI sufficiently addressed the NRC concern.

The NRC staff also expressed concern that the UFSAR update did not include references to the licensee's technical report or the associated NRC evaluation, and requested additional information by e-mail dated December 3, 2020 (ADAMS Accession No. ML20338A151). The licensee's response dated December 17, 2020, added these references.

3.1.5.3 NRC Staff Conclusion

Based on the above, the NRC staff concludes that the licensee has adequately defined the methods used in its analysis such that future calculations will remain consistent with the method approved by the NRC in this LAR and proposed revisions to the UFSAR are sufficient to address NRC concerns. Therefore, the NRC staff concludes the licensee's justification for Limitation and Condition 5 is acceptable.

3.1.6 Limitation and Condition 6

Identify key elements of the risk-informed analysis and corresponding methods, approaches, and data that, if changed, would constitute a departure from the method used in the safety analysis as defined by 10 CFR 50.59.

3.1.6.1 NRC Staff Evaluation and Conclusion

The licensee stated that the key elements of the risk-informed analysis identified in its resolution of Limitation and Condition 5 (see Section 3.1.5 above) are the same as those that are considered to be a method approved by the NRC as defined in 10 CFR 50.59. Based on the NRC staff acceptance of the licensee's response to Limitation and Condition 5, and the licensee's recognition that the key elements are an NRC approved method as defined in 10 CFR 50.59, the NRC staff concludes the licensee's justification for Limitation and Condition 6 is acceptable.

3.1.7 Limitation and Condition 7

Identify the relevant elements of the risk-informed assessment that may need to be periodically updated. The licensee must describe the program or controls that will be used to ensure relevant elements of the risk-informed assessment are periodically updated.

3.1.7.1 NRC Staff Evaluation and Conclusion

The LAR identified relevant elements that will be considered during periodic updates of the risk-informed assessment. The licensee stated that the programs and controls described in its resolution of Limitation and Condition 4 (see Section 3.1.4 above) will be used to ensure the risk-informed assessment is periodically updated. The licensee stated that its periodic updates to the risk-informed assessment will occur within 48 months following initial NRC approval of the LAR or the most recent update. The NRC staff confirmed that, taken in conjunction with the licensee's response to address Limitation and Condition 4, the licensee has identified the set of inputs and attributes that if altered would impact the risk-informed assessment and require an update. Based on the above and NRC staff acceptance of the licensee's response to Limitation and Condition 4, the NRC staff concludes the licensee's justification for Limitation and Condition 7 is acceptable.

3.1.8 Limitation and Condition 8

Describe a reporting and corrective action strategy for addressing situations in which an update to the risk-informed assessment reveals that the acceptance guidelines described in Section 2.4 of RG 1.174, Revision 3, have been exceeded.

3.1.8.1 NRC Staff Evaluation and Conclusion

The staff reviewed the licensee's strategy for reporting non-conforming situations, including exceedance of the Region III acceptance guidelines in RG 1.174, Revision 3, to the NRC and identified a corrective action program for addressing such situations. The NRC staff's review concludes that the licensee's strategy is acceptable because (1) the licensee proposed to use appropriate plant programs and procedures for its corrective action strategy, and (2) the licensee's reporting strategy is consistent with the existing applicable requirements in 10 CFR 50.72 and 10 CFR 50.73. The NRC staff also confirmed that the licensee's proposed UFSAR markup appropriately incorporated the risk-informed assessment acceptance guidelines as RG 1.174, Region III. In Enclosure 2, Attachment 2, of the LAR letter dated August 17, 2020, the licensee provided guidance for supporting operability evaluations of non-conforming conditions, for information, as these programs are covered by other regulatory programs and procedures.

Based on the above, the NRC staff concludes that the licensee provided an adequate reporting and corrective action strategy such that updates to the risk-informed assessment that lead to exceeding RG 1.174, Region III acceptance guidelines will be reported to the NRC and corrected. Therefore, the NRC staff finds that the licensee's justification for Limitation and Condition 8 is acceptable.

3.1.9 Limitation and Condition 9

Correct the error concerning the evaluation of transported coatings debris loads described in SNC's letter dated December 4, 2018. Specifically, provide corrected coating debris volumes and describe how coating debris loads on the strainers are determined. In addition:

a. Verify that the use of the corrected coating debris volumes has a limited impact on strainer head loss and the head loss is acceptable. Also, the licensee must describe the method of verification.

b. Verify that the use of the corrected coating debris volumes has a limited impact on CDF and does not result in exceeding the acceptance guidelines for very small change in risk, as described in Section 2.4 of RG 1.174, Revision 3. Also, the licensee must describe the method of verification.

The NRC staff notes that the correction of the coating error identified in Limitation and Condition 9 is an issue that has the potential to significantly affect the results of the risk-informed analysis. Therefore, it is evaluated in detail in this section.

3.1.9.1 Technical Description of the Coating Debris Load Error

The licensee provided a detailed response to Limitation and Condition 9 in Attachment 3 of Enclosure 2 of its LAR. The licensee stated that the unqualified coatings washdown transport was calculated non-conservatively in the July 2018 submittal. Therefore, the NARWHAL (i.e., the program used to calculate risk) calculations for the risk quantification, and sensitivity and uncertainty analyses were affected and were redone for the LAR submittal.

The licensee stated that the change in the coatings amount (i.e., quantities) required a recalculation of most aspects of the analysis. The outcome is an updated calculation of risk including a reanalysis of the breaks that can result in a failure of one or more acceptance criteria. Therefore, all calculations downstream of identification of break locations and sizes are affected. However, debris generation and transport calculations for materials other than coatings were unaffected.

The licensee provided updated tables of limiting breaks. The tables included debris amounts for four breaks that generate the most debris but do not fail any of the analysis acceptance criteria, and for four breaks that generate the most debris and result in failure of at least one acceptance criteria. The break locations and sizes for the worst-case break that resulted in failures did not change. However, the coating amounts for these breaks increased as expected due to an increased coatings debris term. Debris amounts for the overall worst-case breaks were consistent with those from the submittal dated July 10, 2018, except that the coatings amounts were greater.

The worst-case no failure break locations changed from the licensee's submittal dated July 10, 2018. The updated breaks were larger but transported lower amounts of fiber than the original breaks. The amounts of fibrous debris transported to the Containment Spray (CS) sump strainers were significantly reduced and the transport amounts of other debris types to the CS sump strainers were reduced to zero. The debris amounts transported to the RHR strainers were much lower for fiber, much higher for coatings, greater for calcium phosphate, and lower for sodium aluminum silicate.

The transported debris quantities calculated for the worst-case breaks that did not result in failure were less than expected. In addition, it was not clear why the new limiting break sizes would be larger than the previous limiting break sizes simply due to particulate debris amount increases. By e-mail dated December 3, 2020, the NRC staff requested that the licensee provide information regarding how the debris quantities for these breaks were calculated for the LAR. The NRC staff also requested that the licensee provide a comparison between Table 3.b.4-2 from the July 10, 2018 submittal and Table A3-3 from the LAR. The licensee responded in letter dated December 17, 2020.

In its RAI response, the licensee stated that the NARWHAL models in the LAR were revised to correct washdown fractions for coatings. The licensee also stated that the change did not affect other aspects of the calculational methods for debris generation and transport used in its technical report. The response stated that the limiting fiber breaks that did not fail any acceptance criteria in the 2018 technical report were on the hot leg. These breaks were assumed to cause containment spray (CS) to actuate and resulted in fiber transport just under the limit. The increased coatings washdown fractions used in the LAR resulted in these (and other) breaks failing the strainer coating criteria, so they no longer remain in the table for breaks that do not cause a failure. Previously, the coatings limit was not challenged because of the erroneously low washdown fraction assumed. Some breaks that did not fail any criteria in the technical report fail the coatings criterion in the LAR due to the corrected coatings washdown fractions. The additional failures result in increases in change core damage frequency (ΔCDF) and change large early release frequency (ΔLERF).

The model assumes that CS does not start for cold-leg breaks. In the LAR, the washdown fraction for coatings in cases where CS did not start was also corrected. The washdown of coatings when CS does not start is reduced significantly compared to cases where CS actuates. Based on corrected washdown metrics, the LAR demonstrated that additional failures of the coating acceptance limit occur. Even with the smaller coating washdown fractions for cold-leg breaks, the coatings limit is the dominant failure criterion for some breaks that were not challenged previously. Because of this, fiber amounts for the limiting breaks do not approach the fiber acceptance criterion. Transport of all debris types to the CS strainers was reduced or eliminated because the CS was assumed not to start for the new limiting fiber breaks.

In its RAI response dated December 17, 2020, the licensee also stated that the new bounding breaks have lower chemical debris amounts generated because CS does not actuate for the breaks.

The licensee recalculated the conditional failure probabilities and change in risk associated with debris effects considering the increased particulate debris amounts. These results were provided in Tables A3-7 and A3-8 of the LAR dated August 17, 2020. The values for ΔCDF and ΔLERF increased slightly but remained well within the RG 1.174 range for very small changes.

The licensee reperformed the sensitivity and uncertainty analyses considering the increased particulate debris amounts. The new cases have slightly different results due to the updated coatings washdown fractions. These results were provided in Tables A3-9, A3-10, A3-11 and Figures A3-1, A3-2, A3-3, A3-4 of the LAR. The licensee stated that these analyses continue to support the conclusion that the risk impact of the change is small and confidence in this conclusion remains high.

3.1.9.2 NRC Staff Evaluation

The NRC staff evaluated the licensee’s updated calculations of debris loads, bounding breaks, and risk metrics associated with the change in coatings washdown modeling.

The NRC staff review confirmed that the methodology used to perform the updated calculations was not changed from the previously approved methodology. In its RAI response, dated December 3, 2020, the NRC staff identified that the debris amount and break sizes provided in the LAR and compared the same information provided in the technical report. However, the RAI response dated December 17, 2020, provided a detailed explanation for the differences and confirmed that, with the exception of the washdown fraction correction, the methodology was unchanged. The NRC staff review of the RAI response found the changes in break sizes and debris amounts to be acceptable based on the changes in washdown fractions and corrected coating debris transport amounts.

Several parameters in the model can significantly affect the results. The NRC staff considered how the model calculates results for different scenarios. For example, only hot-leg breaks greater than 15 inches are assumed to result in CSS initiation. If the break occurs on the cold leg, the CSS is not modeled to start, and little to no debris would transport to the CSS strainers. Additionally, the washdown fractions for coatings and fibrous debris are decreased compared to cases where the CSS is running. With the corrected washdown fractions, the coatings transport is greater for both the hot and cold-leg breaks, but the breaks that start CS (large hot-leg breaks) incur a greater increase in transport. The increased coating transport causes LAR cases to calculate failures for breaks that the technical report calculated as successes. These failures are caused by exceeding the coatings debris quantity limit which was not challenged in the technical report cases. With the updated washdown fractions, the coatings debris becomes a more dominant cause of failures.

Because the LAR breaks are on the cold leg rather than the hot leg, the lack of debris transport to the CS strainers is acceptable because the CSS does not actuate. A small amount of debris is assumed to transport to the CS strainers during washdown, but no additional debris reaches these strainers. Transport to the RHR strainers is increased for the cases where CS does not start because the CS strainers do not remove debris from the recirculation pool.

The RAI response also stated that the new bounding breaks have lower chemical debris amounts generated because CS does not actuate for the breaks. The reduction in sodium aluminum silicate for the breaks that do not initiate CS results from reduced wetting of containment materials and is consistent with chemical effects modeling accepted by the staff.

The change resulted in very small increases in the ∆CDF and ∆LERF for the base case and a few sensitivity and uncertainty analysis cases. The NRC staff confirmed the LAR updated the systematic risk assessment with the previously approved methodology and that the updated results continued to meet the RG 1.174 Region III acceptance guidelines and is, therefore, acceptable.

3.1.9.3 NRC Staff Conclusion

Based on the above, the NRC staff concludes that the licensee's updated calculations were performed acceptably and used methods that had been previously reviewed and approved by the staff in its evaluation of the licensee's technical report. The only changes made in the calculations were the coatings washdown transport fractions. The fractions used in the NARWHAL calculations were corrected to the values contained in the technical report. The washdown fractions were changed to correct non-conservative values that were used in the calculations for the technical report. The changes resulted in the calculation of additional breaks that fail one or more acceptance criteria and contribute to increases in ∆CDF and ∆LERF. The updated ∆CDF and ∆LERF results remained within the RG 1.174 Region III guidelines.

Accordingly, the NRC staff concludes that the licensee has sufficiently calculated the increase in risk and appropriately evaluated other changes to its technical report that result from increased washdown fractions for coatings. The LAR and RAI response provide sufficient justification to reconcile Limitation and Condition 9 concerning debris volumes and loads. The calculations were performed using methods previously approved by the NRC staff in its SE on the licensee's technical report. The calculations are consistent with NRC staff approved guidance for the effects of debris on LTCC. The results of the calculations continue to meet the staff guidance for risk-informed changes in RG 1.174 Region III. Therefore, the NRC staff finds that the licensee's justification for Limitation and Condition 9 is acceptable.

The NRC staff finds the licensee's proposed risk-informed methodology and evaluation acceptable.

3.2 TS Changes For Implementation of TSTF-567

3.2.1 Proposed Changes to TS 3.5.2, "ECCS - Operating"

The licensee proposed to modify and move SR 3.5.2.7 from TS 3.5.2 to the new containment sump TS. Therefore, the licensee proposed deletion of SR 3.5.2.7.

The new SR 3.6.7.1 does not limit the visual inspection to the suction inlet, trash racks and screens as currently required by the TSs, but instead requires inspection of the entire containment sump system. The containment sump system consists of the containment drainage flow paths, any design features upstream of the containment sump that are credited in the containment debris analysis, the containment sump strainers (or screens), the pump suction trash racks, and the inlet to the ECCS and CSS piping.

The NRC staff concludes the proposed change is acceptable since the existing requirements are either unchanged or expanded and continue to ensure the containment sump is unrestricted (i.e., unobstructed) and stays in proper operating condition. The proposed change meets the requirements of 10 CFR 50.36(c)(3) because it provides an SR to assure the necessary quality of systems and components are maintained, that facility operation will be within safety limits, and that the LCOs will be met. The NRC staff finds the change to TS 3.5.2 acceptable.

3.2.2 Proposed Changes to TS 3.5.3, "ECCS - Shutdown"

The licensee proposed to delete the reference to SR 3.5.2.7 in SR 3.5.3.1.

The NRC staff concludes the proposed change is acceptable since SR 3.5.2.7 was modified and moved to the new containment sump TS. The existing SR on the containment sump is augmented (by requiring inspection of additional sump components) and moved to the new specification, and a duplicative requirement to perform the SR in TS 3.5.3 is removed. The new specification retains or expands the existing requirements on the containment sump and the actions to be taken when the containment sump is inoperable with the exception of adding new actions to be taken when the containment sump is inoperable due to containment accident generated and transported debris exceeding the analyzed limits. The new action provides time to evaluate and correct the condition instead of requiring an immediate plant shutdown. The proposed change meets the requirements of 10 CFR 50.36(c)(3) because it provides SRs to assure the necessary quality of systems and components are maintained, that facility operation will be within safety limits, and that the LCOs will be met. The NRC staff finds the change to TS 3.5.3 acceptable.

3.2.3 Proposed Addition of Containment Sump TS

3.2.3.1 Evaluation of the Proposed TS

The licensee proposed to add a new TS to address operability requirements of the containment sump. The numbering for this proposed TS is TS 3.6.7.

The containment sump supports the post-accident operation of the ECCS and CSS. However, only the current ECCS TSs contain SRs related to the containment sump and the TSs do not specify required actions that specifically address an inoperable containment sump. If the containment sump (an ECCS and CSS support system) was found to be inoperable, those respective LCOs would not be met. In order to address concerns related to containment sump operability due to debris accumulation described in GSI-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," the licensee proposed to add a new specification to address containment sump inoperability and create a condition for when the sump is inoperable due to analyzed containment accident generated and transported debris.

Based on the below evaluation, the NRC staff determined that the proposed TS satisfies the requirements of 10 CFR 50.36(c)(2)(i) because the LCO specifies the lowest functional capability or performance levels of equipment required for safe operation of the facility. There is reasonable assurance that the required actions to be taken when the LCO is not met can be conducted without endangering the health and safety of the public.

3.2.3.2 Evaluation of the Applicability

The proposed TS requires the containment sump to be operable during Modes 1, 2, 3, and 4. The ECCS and CSS TSs currently are applicable during Modes 1, 2, 3, and 4.

The NRC staff finds the proposed applicability is acceptable because it is consistent with the applicability of the ECCS and CSS TS, which are supported by the containment sump system.

3.2.3.3 Evaluation of Condition A

The licensee has analyzed the susceptibility of the ECCS and CSS to the adverse effects of post-accident debris blockage and operation with debris-laden fluids. The licensee has established limits on the allowable quantities of containment accident generated debris that could be transported to the containment sump based on its current plant configuration. In the current TSs, if unanalyzed debris sources are discovered inside containment, if errors are discovered in debris-related analyses, or if a previously unevaluated phenomenon that can affect containment sump performance is discovered, the containment sump, and the supported ECCS and CSS, may be inoperable and the TSs would require a plant shutdown with no time provided to evaluate the condition.

In order to address this situation and to provide sufficient time to evaluate the condition, the licensee proposed Condition A, which is applicable when the containment sump is inoperable due to containment accident generated and transported debris exceeding the analyzed limits. Under Condition A, the operability of the containment sump with respect to debris is based on a quantity of debris evaluated and determined to be acceptable by the licensee. Conditions not evaluated under Condition A (containment accident generated and transported debris) and that affect the quantity of analyzed debris will be evaluated using a deterministic process.

Under Condition A, Required Action A.1 mandates immediate action to be initiated to mitigate the condition. Although the NRC does not approve the TS Bases, the licensee's proposed TS Bases for Required Action A.1 provided the following examples of mitigating actions:

- Removing the debris source from containment or preventing the debris from being transported to the containment sump;
- Evaluating the debris source against the assumptions in the analysis;
- Deferring maintenance that would affect availability of the affected systems and other LOCA mitigating equipment;
- Deferring maintenance that would affect availability of primary defense-in-depth systems, such as containment coolers;
- Briefing operators on LOCA debris management actions; or
- Applying an alternative method to establish new limits.

The NRC staff finds the proposed Required Action A.1 and its Completion Time (CT) are acceptable because they place immediate urgency on the initiation of the appropriate actions that to mitigate or reduce the impact of the containment accident generated and transported debris for the identified conditions.

Concurrently, Required Action A.2 mandates SR 3.4.13.1, the RCS water inventory balance, to be performed at an increased frequency of once per 24 hours. An unexpected increase in RCS leakage could be indicative of an increased potential for an RCS pipe break, which could result in debris being generated and transported to the containment sump. The NRC staff finds the proposed Required Action A.2 and its CT are acceptable because the more frequent monitoring allows operators to act in a timely manner to minimize the potential for an RCS pipe break while the containment sump is inoperable.

Proposed Required Action A.3 and its CT requires the inoperable containment sump to be restored to operable status in 90 days. The NRC staff finds the proposed Required Action A.3 and its CT are acceptable because they provide a reasonable amount of time for the licensee to diagnose, plan and possibly reduce the severity of, or mitigate the unanalyzed debris condition and prevent a loss of ECCS and CSS safety function. In addition, 90 days is adequate given the conservatisms in the containment debris analysis and the proposed compensatory actions required to be implemented immediately by Required Action A.1. Also, as discussed later in this SE section, the new SR will require visual inspection of the containment sump system (including the containment drainage flow paths, any design features upstream of the containment sump that are credited in the containment debris analysis, the containment sump strainers, the pump suction trash racks, and the inlet to the ECCS and CSS piping for evidence of structural degradation, potential for debris bypass, and presence of corrosion or debris blockage) to ensure no loose debris is present and there is no evidence of structural distress or abnormal corrosion.

For Condition A, a plant with multiple sumps is treated equivalently to a plant with a single sump, because multiple sumps are considered part of a single support system. Vogtle, Units 1 and 2, has four independent sumps. Accident generated debris can affect all the sumps simultaneously so for the purposes of Condition A, the four Vogtle sumps are treated as a single support system.

As detailed above, the NRC staff finds the proposed Required Actions A.1, A.2, and A.3 and its CTs are acceptable because they place urgency on the initiation of the appropriate actions that could mitigate or reduce the impact of the identified conditions.

3.2.3.4 Evaluation of Condition B

Condition B specifies the required actions for when the containment sump is inoperable for reasons other than Condition A. for containment accident generated and transported debris exceeding the analyzed limits. As described in Section 2.4.4, the licensee proposed a variation from the TSTF for the Required Actions associated with Condition B. This section describes the licensee specific implementation of this condition and evaluates its acceptability.

Required Action B.1 requires declaring any ECCS train(s) associated with an inoperable sump inoperable immediately except if the sump inoperability is due to accident generated debris. The Notes from TSTF-567 are not required and are eliminated as described above. Direct, immediate entry into the conditions and required actions for the supported system (ECCS) made inoperable by the associated sump(s) places the plant in a condition consistent with the loss of ECCS function. Since Required Action B.1 directs immediate entry into the applicable ECCS TS for reasons other than containment accident generated and transported debris exceeding the analyzed limits, actions consistent with the operability status of the affected ECCS train(s) are required by the TS. Inoperability due to containment accident generated and transported debris are addressed by Condition A discussed above.

The licensee's proposal also adds Required Action B.2 that requires any CSS trains affected by inoperable sumps for reasons other than containment accident generated debris to be declared inoperable immediately. Direct, immediate entry into the conditions and required actions for the supported system (CSS) made inoperable by the associated sump(s) places the plant in a condition consistent with the loss of CSS function. Since Required Action B.2 directs immediate entry into the applicable CCS TS for reasons other than containment accident generated and transported debris exceeding the analyzed limits, actions consistent with the operability status of the affected CSS train(s) are required by the TS. Inoperability due to containment accident generated and transported debris are addressed by Condition A discussed above.

The proposed CTs for Required Action B.1 and B.2 are "immediately." These CTs are consistent with the loss of ECCS or CSS operability so that the ECCS and CSS TS Required Actions direct the licensee's immediate response and is, therefore, acceptable.

The NRC staff finds the proposed changes are acceptable since they continue to provide remedial actions for conditions where the containment sumps are inoperable for reasons other than Condition A and ensure safe operation of the plant. In addition, the proposed CT is acceptable since it requires the licensee to immediately place the plant in a condition that reflects the inoperability of the affected system(s) and train(s).

3.2.3.5 Evaluation of Condition C

The licensee also proposed a variation from TSTF-567 to Condition C. In the Traveler, Condition C is entered to bring the unit to a mode outside the applicability of the TS if the Required Actions and CTs for Conditions A or B are not met. In the licensee's proposal, Condition B is not included in the Condition C applicability because, in the licensee's variation, Condition B requires immediate entry into LCOs for inoperable supported train(s) of ECCS and CSS. Therefore, it is appropriate to exclude Condition B in the licensee's proposed implementation. For Vogtle, if operators are unable to restore the affected containment sump to operable status under Condition A, Required Action C.1 directs the unit be in Mode 3 in 6 hours, followed by Mode 5 in 36 hours, as required by Required Action C.2.

The NRC staff finds the proposed Condition and its Required Actions are acceptable, in part, because the Condition is consistent with the STS, and the Required Action requires the operators to place the unit in a condition in which the LCO no longer applies. In addition, the proposed CTs allow a reasonable amount of time to decrease from full power conditions to the required plant conditions in an orderly manner and without challenging plant systems.

3.2.3.6 Evaluation of the New SR

The licensee proposed a new SR in the new containment sump TS. This SR is currently located in TS 3.5.2 and referred to in TS 3.5.3. The licensee proposes that the numbering for this new SR be SR 3.6.7.1. The frequency of the proposed SR is in accordance with the SFCP.

The proposed SR requires verification, by visual inspection, that the containment sump does not show structural damage, abnormal corrosion, or debris blockage.

The proposed SR is stated in generic terms and expands the scope of the required visual inspection to include the entire containment sump system. The entire containment sump system consists of the containment drainage flow paths, the containment sump strainers (or screens), the pump suction trash racks, and the inlets to the ECCS and CSS piping.

The NRC staff finds the proposed new SR is acceptable since it expands the scope of inspection of the original SR. In addition, the proposed frequency is acceptable since it is the same as that currently required by the TSs. Therefore, the NRC staff finds that, as required by 10 CFR 50.36(c)(3), the necessary quality of systems will be maintained in accordance with the associated LCOs. The NRC staff finds the addition of TS 3.6.7 acceptable and would meet 10 CFR 50.36(c)(3).

3.2.3.7 Evaluation of Changes to Table of Contents

The licensee also proposed a conforming change to the Table of Contents to include the new containment sump TS. This conforming change is acceptable since it is editorial in nature and supports the inclusion of the new containment sump TS. The NRC staff finds the changes to the Table of Contents acceptable.

3.2.3.8 Evaluation of Changes to the TS Bases

The licensee submitted TS Bases changes (that corresponded to the proposed TS changes) to provide the reasons for the proposed TSs. The licensee stated that the TS Bases changes are consistent with the Bases changes in the model application. The NRC staff does not approve TS Bases.

3.2.3.9 Conclusion Regarding Proposed Containment Sump TS

The new containment sump TS retains and expands the existing TS requirements except the addition of Condition A. Condition A provides a condition for an inoperable containment sump due to containment accident generated and transported debris exceeding the analyzed limits.

The NRC staff reviewed the proposed changes against the regulations and concludes that, for the reasons discussed above, the changes continue to meet the requirements of 10 CFR 50.36(c)(2)(i) and 50.36(c)(3) and thus provide reasonable assurance of adequate protection. The revised TSs will have the requisite requirements and controls to ensure continued safe operation and controls for sump operability for Vogtle, Units 1 and 2. Therefore, the NRC staff concludes that the proposed new TS 3.6.7 is acceptable.

3.2.4 Variations

The Vogtle TSs utilize different numbering than the STS on which TSTF-567 was based. Specifically, SR 3.5.2.8 in NUREG-1431 (Westinghouse STS) is 3.5.2.7 in the Vogtle TS. Additionally, TS 3.6.19 in TSTF-567 is proposed TS 3.6.7 in the Vogtle TS. These numbering differences are editorial and do not affect the applicability of TSTF-567 to the proposed LAR.

As the Vogtle TSs contain a SFCP, the frequency for SR 3.6.7.1 is "[i]n accordance with the Surveillance Frequency Control Program." The SFCP was previously incorporated into the TS and applied to SR 3.5.2.7 that is proposed to be replaced by SR 3.6.7.1. Although the requirements are somewhat expanded, SR 3.6.7.1 will perform the same function as SR 3.5.2.7, and the intent of the proposed SR is the same. Therefore, the NRC staff finds it acceptable to apply the SFCP to the proposed SR 3.6.7.1.

The licensee proposed Required Actions for Condition B different from those specified in TSTF-567. The proposal eliminates the Traveler Notes 1 and 2 that require the affected unit to enter the applicable Conditions and Required Actions of LCO 3.5.2 and 3.5.3, and 3.6.6 for trains made inoperable by the containment sumps. The proposal also changes the CT from the traveler that allows 72 hours (or in accordance with the risk informed completion time) to restore the sumps to operable status. Instead, the licensee proposed Required Actions (B.1 and B.2) to declare the affected ECCS and CSS trains inoperable. The CT for these actions is "immediately." The licensee's proposal requires placing the plant in a condition appropriate for the ECCS and/or CSS rendered inoperable due to sump inoperability by direct entry into the appropriate actions of TS 3.5.2 and 3.5.3, and 3.6.6. This simplification is possible because each CSS and ECCS pump at Vogtle has its own strainer and sump. The NRC staff finds the proposed Required Actions for Condition B are acceptable because they immediately require the licensee to place the plant in a condition equal to or more conservative than the Traveler and simplify the implementation of actions associated with Condition B.

The licensee also proposed a variation to eliminate Condition B from Condition C. Condition C requires that the unit be placed in Mode 3 in 6 hours and Mode 5 in 36 hours if required actions and completion times are not met. In the variation, Condition C is applied only if the Condition A Required Actions are not met within the CT. Elimination of Condition B from Condition C is appropriate because Condition B requires the licensee to declare affected trains of ECCS and CSS inoperable immediately on loss of function due to sump inoperability, except as specified in Condition A, by direct entry into the appropriate conditions of TS 3.5.2, 3.5.3, and 3.6.6. These TSs provide direction to act based on the inoperability of the ECCS and CSS trains affected by sump inoperability. The purpose of Condition C is to require the plant to exit the Modes of applicability for the new sump TS 3.6.7 if the Required Actions for Conditions A and B are not performed within the appropriate CTs. The variation for Condition C still requires this for Condition A. For Condition B, appropriate actions for sump inoperability are required by the TS for ECCS and CSS. Therefore, the NRC staff finds that the proposed variation to Condition C, which makes it applicable only to Condition A, is acceptable.

3.3 TECHNICAL EVALUATION CONCLUSION

The NRC staff determined that the proposed TS changes would continue to meet the standards for TS in 10 CFR 50.36 and are, therefore, acceptable. As required by 10 CFR 50.36(c)(2), the LCOs specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The proposed changes to the SR assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met, and satisfy 10 CFR 50.36(c)(3).

The NRC staff previously reviewed a technical report submitted by the licensee. The NRC found the technical report acceptable for use in plant-specific licensing applications for Vogtle, Units 1 and 2. The NRC staff's conclusions are documented in a final staff evaluation of the Vogtle technical report dated September 30, 2019. The limitations and conditions included in the NRC staff evaluation were the only outstanding technical areas that required additional technical evaluation. The NRC staff finds that the licensee adequately addressed each of the Limitation and Conditions detailed by the NRC staff evaluation of the licensee's technical report. Therefore, the NRC staff concludes that the SNC's proposed risk-informed assessment methodology for assessing the effects of debris on LTCC at Vogtle (including, submodels and integration of the submodels) described in the technical report enclosed with the letter dated July 10, 2018, is an acceptable evaluation model, as required by 10 CFR 50.46, for use in licensing actions concerning LTCC.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, State of Georgia officials were notified of the proposed issuance of the amendment on March 9, 2021. On March 9, 2021, the State official confirmed had State of Georgia has no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding on November 3, 2020 (85 FR 69656). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: July 30, 2021

ATTACHMENT

HISTORICAL INFORMATION AND SYSTEM DESCRIPTIONS

Challenges to Safety Systems Function from Debris in Containment

The function of the emergency core cooling system (ECCS) is to cool the reactor core and provide shutdown capability following a loss-of-coolant accident (LOCA). The primary functions of the containment spray system (CSS) are to reduce containment pressure and reduce the concentration and quantity of fission products in the containment building after a LOCA.

Nuclear plants are designed and licensed with the expectation that they are able to remove reactor decay heat following a LOCA to prevent core damage. Long-term core cooling (LTCC) following a LOCA is also a basic safety function for nuclear reactors. The recirculation sump located in the lower areas of the reactor containment structure provides a water source to the ECCS for extended cooling of the core in a pressurized water reactor (PWR) once the initial water source has been depleted and the systems are switched over to recirculation mode.

If a LOCA occurs, piping thermal insulation and other materials located in containment may be dislodged by the two-phase (steam and liquid) water jet emanating from the broken pipe. This debris may be transported by the flow of water and steam from the break or from the CSS to the pool of water that collects in the containment recirculation sump. Once transported to the sump pool, the debris could be drawn toward the ECCS sump strainers, which are designed to prevent debris from entering the CSS and the ECCS. If this debris clogs the strainers, the ECCS could fail, resulting in core damage, or the CSS pumps could fail, resulting in containment pressure or radiation dose increasing beyond deterministic limits. It is also possible that some debris could bypass the sump strainers and get lodged in the reactor core. This could result in reduced core cooling and potential core damage.

Generic Safety Issue (GSI)-191

In 1996, the U. S. Nuclear Regulatory Commission (NRC) identified an issue associated with the effects of debris accumulation on PWR sump performance during design-basis accidents (i.e., GSI-191). This issue was similar to concerns at boiling-water reactors (BWRs), to new information identified following closure of the actions taken for resolution of the issue at BWRs, and to confirmatory testing conducted by the NRC.

Findings from research and industry operating experience raised questions concerning the adequacy of PWR sump designs. Research findings demonstrated that the amount of debris generated and transported by a high-energy LOCA could be greater than originally anticipated. The debris from a LOCA could also be finer, and thus, more easily transportable, and could be comprised of debris consisting of fibrous material combined with particulate material that could result in a substantially greater flow restriction than an equivalent amount of either type of debris alone. These research findings prompted the NRC to open GSI-191.

The two distinct but related safety concerns are: (1) potential clogging of the sump strainers that results in ECCS or CSS pump failure, and (2) potential clogging of flow channels within the reactor vessel because of debris bypassing the sump strainers, often referred to as in-vessel effects. Clogging at either the strainers or in-vessel channels can result in loss of the LTCC safety function.

More information on the background, testing, and other actions associated with GSI-191 can be found in NUREG-0897, “Containment Emergency Sump Performance: Technical Findings Related to Unresolved Safety Issue A-43,” dated October 1985 (ADAMS Accession No. ML112440046), and NRC Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors,” dated June 9, 2003 (ADAMS Accession No. ML031600259).

Generic Letter 2004-02

As part of the actions to resolve GSI-191, in September 2004, the NRC issued GL 2004-02 (ADAMS Accession No. ML042360586) to holders of operating licenses for PWRs. In GL 2004-02, the NRC staff requested that licensees perform an evaluation of their ECCS and CSS recirculation functions, considering the potential for debris-laden coolant to be circulated by the ECCS and the CSS after a LOCA or high-energy line break inside containment, and, if appropriate, take additional action to ensure system function. GL 2004-02 required, per 10 CFR 50.54(f), that licensees provide the NRC a written response describing the results of their evaluation and any modifications made, or planned, to ensure ECCS and CSS system function during recirculation following a design-basis event, or any alternate action proposed and the basis for its acceptability.

The NRC staff requirements memorandum (SRM) associated with SECY-10-113, “Closure Options for Generic Safety Issue 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance,” dated December 23, 2010 (ADAMS Accession No. ML103570354), directed the NRC staff to consider a risk-informed approach for resolution of GSI-191. In 2012, the NRC staff developed three options to resolve GSI-191. These options were documented and proposed to the Commission in SECY-12-0093 (ADAMS Accession No. ML121310648). The options are summarized as follows:

- Option 1 allows licensees to demonstrate compliance with 10 CFR 50.46 through approved models and test methods.

- Option 2 requires implementation of additional mitigating measures and allows additional time for licensees to resolve issues through further industry testing or use of a risk-informed approach.

- Option 3 involves separating the regulatory treatment of the sump strainer and in-vessel effects so that strainer issues can be treated deterministically and in-vessel issues can be risk-informed.

These options allowed industry alternative approaches for resolving GSI-191. The Commission issued SRM-SECY-12-0093 on December 14, 2012 (ADAMS Accession No. ML12349A378), approving all three options for closure of GSI-191.

By letter dated May 16, 2013 (ADAMS Accession No. ML13137A130), SNC stated that it would pursue Option 2 for the closure of GSI-191 and GL 2004-02, and intended to use a risk-informed methodology.

Description of Affected Structures, Systems, and Components

A fundamental function of the ECCS is to recirculate water that has collected in the containment sump following a break-in the reactor coolant system (RCS) piping to ensure long-term removal of decay heat from the reactor fuel. Leaks from the RCS in excess of the plant's normal makeup capability (scenarios known as LOCAs), are part of a nuclear power plant's design bases. Hence, nuclear plants are designed and licensed with the expectation that they are able to remove reactor decay heat following a LOCA to prevent core damage. Long-term cooling following a LOCA is also a basic safety function for nuclear reactors. The recirculation sump located in the lower areas of the reactor containment structure provides a water source to the ECCS in a PWR once the initial water source has been depleted and the systems are switched over to recirculation mode for extended cooling of the core.

Emergency Core Cooling System

The ECCS consists of the centrifugal charging pumps; safety injection (SI) pumps; residual heat removal (RHR) pumps; accumulators, boron injection tank (Unit 1 only); RHR heat exchangers; refueling water storage tank (RWST); and the associated piping, valves, instrumentation, and other related equipment. The primary function of the ECCS following an accident is to remove the stored and fission product decay heat from the reactor core such that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented.

As stated in the Vogtle Updated Final Safety Analysis Report (UFSAR):

> The ECCS is designed to cool the reactor core and to provide additional shutdown capability following initiation of the following accident conditions:
>
> A. Loss-of-coolant accident (LOCA) including a pipe break or a spurious relief or safety valve opening in the RCS which would result in a discharge larger than that which could be made up by the normal makeup system.
>
> B. Loss-of-secondary-coolant accident including a pipe break or a spurious relief or safety valve opening in the secondary steam system which would result in an uncontrolled steam release or a pipe break-in the secondary feedwater system.
>
> C. A steam generator tube rupture accident.
>
> Emergency core cooling following a LOCA is divided into three phases:
>
> A. Short-Term Core Cooling/Cold Leg Injection Phase
>
> The cold leg injection phase is defined as that period during which borated water is delivered from the RWST and accumulators to the RCS cold legs.
>
> B. Long-Term Core Cooling/Cold Leg Recirculation
>
> The cold leg recirculation phase is that period during which borated water is recirculated from the containment emergency sump to the RCS cold legs.
>
> C. Long-Term Core Cooling/Hot Leg Recirculation Phase
>
> The hot leg recirculation phase is that period during which borated water is recirculated from the containment emergency sump to both the RCS hot legs and RCS cold legs.

In the event of an accident, the RHR pumps are started automatically. The RHR pumps take suction from the RWST during the injection phase and are automatically realigned to the containment emergency sump during the recirculation phase, although manual action is required to close the suction path from the RWST. In the event of an accident, the SI pumps are started automatically on receipt of an SI signal. These pumps deliver water to the RCS from the RWST during the injection phase and from the containment emergency sump via the RHR pumps during the recirculation phase. When a predetermined low RWST level is reached, the SI and charging pumps are manually aligned to take suction from the RHR pump discharge headers.

There are two ECCS or RHR sumps. A screen is installed on each sump. The Vogtle sump strainers or screens consists of stacked disk strainers designed by General Electric. The RHR screens are composed of four stacks of 18 disks. Each stack is 30 inches long by 30 inches wide by about 54 inches high. Four stacks provide 765 square feet (ft^2) of perforated plate area and 179 ft^2 of circumscribed surface area per sump. Each disk is a welded assembly of two perforated plates and their structural support components. The screens are designed to withstand the loading associated with the maximum design values of debris mass and differential pressure. The plate-hole (perforation) diameter of the screen is 3/32 in. A small percentage of the holes are larger than 3/32 in diameter, but none are larger than 1/4-inch diameter. If particles penetrate the screen, they will pass through the downstream piping pumps and valves and may enter the reactor vessel. The effects of debris penetration on downstream equipment were evaluated in the licensee's technical report and the associated staff evaluation. The effects of debris that reach the reactor is evaluated in the LAR and this safety evaluation. The screens bolt to the floor and may be removed by unbolting individual screen sections.

The RWST serves as a source of emergency borated cooling water for injection and containment spray.

Containment Spray System (CSS)

The CSS consists of two pumps, spray ring headers and spray nozzles, valves, and connecting piping. Initially, water from the RWST is used for the containment spray. When the RWST level is low the CSS switches to use water recirculated from the containment emergency sump. The recirculated spray is mixed with trisodium phosphate in the containment sump region.

There are two containment spray (CS) sumps. A screen is installed on each sump. The CS screens are composed of four stacks of 14 disks. Each stack is 30 inches long by 30 inches wide by about 42 inches high. Four stacks provide 590 ft^2 of perforated plate area and 139 ft^2 of circumscribed surface area per sump. The design attributes of the CS screens are similar to the RHR screens described above. If particles penetrate the screen, they will pass through the piping pumps and valves, as well as the 3/8-inch diameter CS nozzle openings, without difficulty.